UCRL- 98173
PREPRINT

PROBABILISTIC TREATMENT OF STRESS CORRROSION CRACKING

Garry S. Holman

This Paper Was Prepared for Submittal to the
NRC-MITI Technical Information Exchange
Engineering Workshop

Tokyo, Japan / May 16-20, 1988

May 18, 1988

Lawrence Livermore National Laboratory

PROBABILISTIC TREATMENT OF STRESS CORROSION CRACKING*

Garry S. Holman

Lawrence Livermore National Laboratory
University of California
Livermore, California / U.S.A.

ABSTRACT

The Lawrence Livermore National Laboratory (LLNL) has estimated the probability of double-ended guillotine break (DEGB) in the reactor coolant piping of Mark I boiling water reactor (BWR) plants. Two causes of pipe break are considered: crack growth at welded joints and the seismically-induced failure of component supports. For the former a probabilistic fracture mechanics model is used, for the latter a probabilistic support reliability model. This paper describes a probabilistic model developed to account for effects of intergranular stress corrosion cracking (IGSCC). The IGSCC model, based on experimental and field data compiled from several sources, correlates times to crack initiation and crack growth rates for Types 304 and 316NG stainless steel against material-specific "damage parameters" which consolidate the separate effects of coolant environment (temperature, dissolved oxygen content, level of impurities), stress (including residual stress), and degree of sensitization. Application of this model to actual BWR recirculation piping shows that IGSCC clearly dominates the probability of failure in 304SS piping, mainly due to cracks that initiate within a few years after plant operation has begun. Replacing Type 304 piping with 316NG reduces failure probabilities by several orders of magnitude.

1. INTRODUCTION

The Lawrence Livermore National Laboratory (LLNL), through its Nuclear Systems Safety Program, has performed probabilistic reliability analyses of PWR and BWR reactor coolant piping for the NRC Office of Nuclear Regulatory Research. Specifically, LLNL has estimated the probability of a double-ended guillotine break (DEGB) in the reactor coolant loop piping of PWR plants, and in the main steam, feedwater, and recirculation piping of BWR plants. For these piping systems, the results of these investigations provide NRC with one technical basis on which to:

* "This work was supported by the United States Nuclear Regulatory Commission under a Memorandum of Understanding with the United States Department of Energy."

(1) reevaluate the current general design requirement that DEGB be postulated in the design of nuclear power plant structures, systems, and components against the effects of postulated pipe breaks. Recent NRC rulemaking actions, based in large part on the results of our PWR evaluations, now provide a means for eliminating dynamic effects of reactor coolant loop breaks (e.g., pipe whip, jet impingement) as a basis for PWR plant design.

(2) determine if an earthquake could induce a DEGB and thus reevaluate the design requirements that pipe break loads be combined with those resulting from a safe shutdown earthquake (SSE). Recent deviations from the NRC Standard Review Plan, for example, now allow decoupling of SSE and DEGB loads for PWR reactor coolant loop piping.

(3) make licensing decisions concerning the replacement, upgrading, or redesign of piping systems, or addressing such issues as the need for pipe whip restraints on reactor coolant piping.

In estimating the probability of DEGB, LLNL considers two causes of pipe break; pipe fracture due to the growth of cracks at welded joints ("direct" DEGB) and pipe rupture indirectly caused by the seismically-induced failure of critical supports or equipment ("indirect" DEGB).

2. BACKGROUND

Over the past several years, generic evaluations of reactor coolant loop piping were completed for PWR nuclear steam supply systems manufactured by Westinghouse, Combustion Engineering, and Babcock & Wilcox. In these evaluations, LLNL performed the following:

(1) estimated the probability of direct DEGB taking into account such contributing factors as the initial size (depth and length) of pre-existing fabrication flaws, pipe stresses due to normal operation and sudden extreme loads (such as earthquakes), the crack growth characteristics of pipe materials, and the capability to detect cracks or to detect a leak if a crack were to penetrate the pipe wall. For this purpose, LLNL developed a probabilistic fracture mechanics model using Monte Carlo simulation techniques, implemented in the PRAISE (**P**iping **R**eliability **A**nalysis **I**ncluding **S**eismic **E**vents) computer code.

(2) estimated the probability of indirect DEGB by identifying critical supports or equipment whose failure could result in pipe break, determining the seismic "fragility" (relationship between seismic response and probability of failure) of each, and then combining this result with the probability that an earthquake occurs producing a certain level of excitation ("seismic hazard").

(3) for both causes of DEGB, performed sensitivity studies to identify key parameters affecting the probability of pipe break. We also performed uncertainty studies to quantify how uncertainties in

input data affect the uncertainty in the final estimated probability of pipe break.

The results of these evaluations consistently indicated that the probability of a DEGB in PWR reactor coolant loop piping is extremely small, about 1E-7 events per reactor-year from indirect causes, and less than 1E-10 events per reactor-year from direct causes. It was also found that thermal stresses dominated the probability of direct DEGB, and that earthquakes contributed only negligibly. These results suggested that the DEGB design requirement — and with it related design issues such as coupling of DEGB and SSE loads, asymmetric blowdown, and the need to install pipe whip restraints — warranted reevaluation for PWR reactor coolant loop piping. Details of these investigations have been extensively documented elsewhere [1,2,3,4] and will not be discussed here any further.

The objectives and approach of the BWR study [5] were essentially the same except that additional potential failure mechanisms were added. LLNL limited its investigation to Mark I plants, which have recirculation piping particularly susceptible to the effects of intergranular stress corrosion cracking (IGSCC). Although our evaluations were all generally similar, two important aspects distinguished the BWR study from the earlier PWR evaluations:

(1) the susceptibility of certain BWR stainless steels to stress corrosion cracking required development of an appropriate probabilistic model of corrosion phenomena. Stress corrosion is generally not perceived as a problem in PWR primary loop piping and, as a result, was not considered in our earlier evaluations.

(2) the greater complexity and flexibility of BWR recirculation piping compared to PWR primary loops required that intermediate pipe supports (e.g., snubbers, spring hangers) be incorporated in the evaluation.

This paper will focus on the methods we developed to probabilistically model stress corrosion cracking and on application of the model to a "representative" BWR recirculation system (Fig. 1). The methods used to address failure of intermediate supports are described in detail in a companion paper [6] and will not be discussed here.

3. DOUBLE-ENDED GUILLOTINE BREAK CAUSED BY CRACK GROWTH

The probability of "direct" DEGB in reactor coolant piping is estimated using a probabilistic fracture mechanics model implemented in the PRAISE computer code and associated pre- and post-processing routines. Details of the model are documented elsewhere [7,8] and will not be repeated here, but can be summarized as follows.

For a given weld joint in a piping system, the probability of failure (i.e., leak or break) is estimated using a Monte Carlo simulation technique. As diagramed in Fig. 2, each replication of the simulation — of which a typical simulation may include many thousand — begins with a pre-existing flaw having initial length and depth randomly selected from appropriate distributions. These distributions in turn relate the probability of crack existence. Fatigue crack growth is then calculated using a Paris growth model, to which are applied stresses associated with normal operating conditions and postulated seismic events. The influence of such factors as non-destructive examination (NDE) and leak detection is also considered through the inclusion of appropriate statistical distributions (e.g., probability of crack non-detection as a function of crack size). Leak occurs when a crack grows through the pipe wall, break when failure criteria based on net section stress (for austenitic materials) or tearing modulus (for carbon steels) are exceeded.

Completing all replications for a given weld joint and tabulating those cracks that cause failure yields the cumulative probability of failure as a function of time at that weld. If only pre-existing cracks are considered, then "stratified sampling" can be applied to assure that initial crack samples are selected only from those sizes that can potentially cause pipe break. Through this technique, very low failure probabilities (less than one in a million) can be reliably estimated from only a few thousand replications of the Monte Carlo simulation.

After the failure probabilities at all weld joints in a piping system have been estimated, a "systems analysis" combines these results with the non-conditional crack existence probability (a function of total volume of weld material) and seismic hazard (which relates the occurrence rates of earthquakes as a function of peak ground acceleration) to obtain the non-conditional probabilities of leak and DEGB.

This was the basic approach followed in our evaluations of PWR reactor coolant loop piping. One significant factor complicating the evaluation of BWR piping, however, was the need to include effects of intergranular stress corrosion cracking (IGSCC). When present, IGSCC not only accelerates the growth rate of existing flaws, but also causes new cracks to initiate after plant operation has begun. The effect of these "initiated" cracks on the probability of DEGB must be therefore be considered in addition to that of pre-existing flaws.

Recirculation piping in older BWR plants, particularly those characterized by the General Electric Mark I containment design, has been found in recent years to be susceptible to intergranular stress corrosion cracking. Stress corrosion cracking occurs in stainless steel piping (in the Mark I plants, Type 304) when the "appropriate" (in an adverse sense) conditions of "sensitization" — material properties conducive to IGSCC that result from prolonged exposure to high temperatures during welding — environment, and stress are met. Earlier versions of PRAISE treated the effect of IGSCC on pre-existing cracks through a simple relationship between growth rate and the stress intensity factor at the crack front; crack initiation was not modeled at all. It is important to note that this model was not applied in our

PWR evaluations because operating experience has indicated that IGSCC is not a problem in PWR reactor coolant loop piping.

4. PROBABILISTIC MODEL OF STRESS CORROSION CRACKING

As part of our BWR study we developed an advanced IGSCC model for the PRAISE code [9]. This model is semi-empirical in nature, and is based on experimental and field data compiled from several sources. Using probabilistic techniques, the model addresses the following IGSCC phenomena:

- **crack initiation**, including the effects of environment, applied loads, and material type (i.e., sensitization). Crack location, time of initiation, and velocity upon initiation are all defined by appropriate distributions based on experimental data.

  "Initiated" cracks are considered separately from pre-existing cracks until one of the following two criteria are satisfied: (1) the crack attains a depth of 0.1 inch, or (2) the velocity of the crack estimated according to the Paris growth law exceed the initiation velocity. Beyond this point, "initiated" and "fracture mechanics" cracks are treated identically.

- **crack growth rate**, including effects of environment, applied loads, and material type.

- **multiple cracks**. Because our earlier evaluations were based on pre-existing flaws only, each Monte Carlo replication included one crack only. Inclusion of crack initiation requires that multiple cracks be considered during each replication.

- **crack linking**. Treating multiple cracks requires that their potential linkage into larger cracks be considered. This is done using linkage criteria specified in Section XI of the ASME Boiler and Pressure Vessel Code.

The model covers not only the Type 304 stainless steel (304SS) found in most Mark I recirculation piping, but Type 316NG "nuclear grade" steel as well, a low-carbon alloy widely regarded as an IGSCC-resistant replacement for Type 304. Crack growth rates and times-to-initiation for each material are correlated against "damage parameters" which consolidate the separate influences of several individual parameters. The damage parameters are multiplicative relationships among exponential terms which individually describe the effects of the various phenomena on IGSCC behavior, including:

- **environment**, specifically coolant temperature, dissolved oxygen content, and level of impurities.

- **applied loads**, including both constant and variable loads to account for steady-state operation and plant loading or unloading, respectively.

- **residual stresses.** Steady-state pipe loads due to welding residual stresses are considered in addition to fatigue loads.

- **material sensitization.**

Figures 3 and 4 show, respectively, times-to-initiation and crack growth rates for 304SS, the material on which the initial development of the model was based. The solid curved lines in Fig. 4 show crack growth rates predicted by the earlier IGSCC model in PRAISE for oxygen concentrations of 0.2 ppm (typical during plant operation) and 8 ppm (typical during startup); the relatively close agreement implies that the earlier model gave reasonable crack growth rates despite its much simpler approach.

The damage parameters in the 304SS model were based on the results of both constant-load (CL) and constant extension rate (CERT) IGSCC laboratory tests. Many other factors were considered during initial model development, but were later excluded from consideration either because they were judged to be of secondary influence for 304SS, or because suitable operating data was not available to exercise them in a plant-specific evaluation. The model also assumes that growth rates and times-to-initiation measured under intentionally harsh laboratory conditions can be extrapolated to the relatively benign conditions found in actual reactors. We regarded this assumption as conservative, noting, for example, that some experimental observations [10] suggest levels of stress intensity factor below which stress corrosion cracking is effectively arrested or at least significantly reduced. Our original simplified model of IGSCC allowed for such "threshold" behavior (see Fig. 5), the present advanced model does not.

Although the present model was developed for 304SS, adapting the correlation scheme for 316NG was a relatively straightforward matter of defining new damage parameters based on appropriate laboratory data; the basic functional form of the model was otherwise left unchanged. Two features unique to the 316NG model are, however, noteworthy:

- where both CERT and CL data were available for 304SS, only CERT data was available for 316NG. These data were used to define constant-load growth rates and times-to-initiation in 316NG under the assumption that the creep behavior of both alloys is similar.

- as noted earlier, three conditions are necessary for IGSCC in austenitic steels: stress, environment, and sensitization. In 304SS, whenever stress corrosion cracking occurs in laboratory tests intended to simulate operating BWR conditions, it is most often intergranular. In 316NG, however, CERT specimens fail by transgranular stress corrosion cracking (TGSCC), whereas IGSCC is observed in fracture mechanics specimens. Since the relative influence of environment and loading on TGSCC in 316NG appears similar to that of IGSCC in 304SS, the available TGSCC data were used to predict cracking in 316NG.

Residual stresses are treated as a random variable in the Monte Carlo simulation. Distributions of residual stress as a function of distance from the inner pipe wall were developed from experimental data for three categories of nominal pipe diameter. For large lines (20 to 26 inches), residual stresses took the form of a damped cosine through the wall as based on data collected by General Electric and Argonne National Laboratory (see Fig. 5). The nominal tensile stress at the inner pipe wall is about 40 ksi. For intermediate-diameter (10 to 20 inches) and small-diameter (less than 10 inches) lines, a linear distribution was assumed through the pipe wall with respective inside wall stresses of 9.3 ksi and 24.4 ksi.

The 304SS model was benchmarked by comparing predicted leak rates under nominal BWR applied load conditions against actual leak and crack indication data made available to us by the NRC Office of Nuclear Reactor Regulation (NRR). During benchmarking we quickly ascertained that residual stress was the parameter most influencing the predicted leak rates, and we therefore opted to "tune" the model on this basis. A variety of schemes were considered before we settled on adjusting the stress magnitude (using a multiplication factor) to bring the model into agreement with the field data. Figure 6 compares predicted leak rates against field data for various adjustment factors, Figure 7 the number of NDE indications with depth a greater than 10 and 50 percent of the wall thickness h, based on the optimum stress adjustment factor. As Fig. 6 shows, surprisingly large reduction factors had to be applied to bring the model into line with the field data, suggesting that factors other than residual stress may be more influential than we first concluded.

Calculations performed during final development of the 316NG model revealed several interesting characteristics of its behavior compared to that of the less-resistant 304SS. For example, we performed analyses both for initiated cracks and for pre-existing cracks, the latter case reflecting only the effect of stress corrosion on crack growth and not only the addition of new "initiated" cracks to the overall population. Figure 8 shows a typical set of results from these analyses, in this case cumulative leak probabilities for an intermediate-diameter weld. Two observations are significant here:

- at any given time, the estimated failure probability in 304SS is some two to three orders of magnitude higher than in 316NG.
- the time required to reach a given leak probability is about six times as long in 316NG as it is in 304SS.

These results also show that where failure in 304SS is always dominated by initiated cracks (i.e., resulting from stress corrosion), in 316NG the initiated cracks dominate the probability of leak only after about 12 years. Once cracks are present, however, growth rates are nominally the same in either material. Consequently, the predicted difference in behavior between the two materials is due to differences in the times-to-initiation and in the number of initiated cracks, rather than differences in their "fracture mechanics" characteristics.

5. PROBABILITY OF FAILURE IN BWR RECIRCULATION LOOP PIPING

After we completed development of the stress corrosion model, we applied it to the recirculation loop piping in an actual Mark I BWR plant. We estimated the leak and DEGB probabilities both for an existing recirculation loop (Fig. 1), and for a proposed "replacement" loop (Fig. 9) fabricated from 316NG. Aside from its use of the more corrosion-resistant material, the replacement loop differs from the original by having fewer weld joints (about 30 compared to 50) and by eliminating entirely the pump bypass line (see Table 1).

During development of the IGSCC model, we found that its complexity greatly increased computer time requirements for its execution (up to three CPU hours per weld for the 20000 to 50000 Monte Carlo replications typical of our analyses) compared to our earlier PWR reactor coolant loop assessments. In order to keep the computational effort within practical bounds, we grouped the welds in the BWR pilot plant recirculation piping, taking those welds with the highest applied loads in each group. We then estimated the leak and DEGB probabilities at each of these representative welds and performed a systems analysis assuming that these leak and DEGB probabilities applied to all welds in the respective group. We followed a similar procedure for the proposed replacement system.

Practical considerations aside, the assumption of "worst case" stress conditions for each weld group offers reasonable assurance that the results of the analysis will be conservative. This conservatism is further enhanced by the fact that we did not include in-service inspection (ISI) in our evaluations (although PRAISE has this capability), nor did we consider how such IGSCC mitigating measures as weld overlay or inductive heating stress improvement (IHSI) might influence the estimated failure probabilities. Our main objective was to investigate the relative behavior of different material types under otherwise nominally identical conditions.

Figures 10(a) and 10(b) show, respectively, cumulative per-loop system leak and DEGB probabilities estimated by PRAISE for the existing loop configuration (i.e. including bypass piping). Results are given both for the original 304SS material and for the Type 316 nuclear grade. In the 304SS piping, leak is predicted to occur after about ten years of operation (i.e. the cumulative probability of leak approaches one). While it is important to keep in mind the conservatism of the analysis, this result is nonetheless reasonably consistent with some field observations. The corresponding probability of DEGB is on the order of 1E-2 after 10 years (or about 1E-3 per year), increasing only slightly (by about a factor of two) over the remaining 30 years of plant life.

If the 304SS is replaced with 316NG while keeping the original piping configuration (a fictitious intermediate step between the existing loops in our pilot plant and the replacement system actually proposed), corresponding leak and break probabilities are nominally zero after 10 years of operation. The probability of leak first exceeds 1E-4 after about 12 years, increasing to about 5E-1 at the end of plant life. Two DEGB events (out of 25000 Monte Carlo replications) were predicted in the riser weld, the first of which occurred at about 30 years; all other weld groups experienced no DEGB events over the entire 40 years of plant life. The resultant end-of-life system break probability is about 2E-3 per loop, or about 2E-4 per loop-year; keep in mind that this result assumes (1) no "threshold" behavior, (2) no ISI over the 30-year period, (3) worst-case applied stresses, and that (4) all risers in the system behave identically. For the "replacement" loop configuration actually proposed, the end-of-life DEGB probability falls to about 1E-3 per loop (1E-4 per loop-year), due to fewer welds in the new system (Fig. 11).

The bar charts in Fig. 12 show the relative contribution each weld type makes to the overall system probabilities of leak and DEGB; note that Fig. 12 does not depict the number of predicted failures, which were far fewer in the 316NG material than in the 304SS. In the existing loop configuration, about 80 percent and 20 percent of the breaks, and about 65 percent and 25 percent of the leaks, occurred at riser welds and bypass line welds, respectively. The remaining leaks predicted (about 10 percent of the total) were distributed, in descending order, among header, discharge line, and suction line welds.

In the proposed replacement system, virtually all leaks occurred in riser welds. System break resulted solely from riser DEGB as discussed above, which Fig. 12(b) reflects.

The relative contribution of different weld types is further illustrated by Fig. 13, which shows weld-by-weld leak probabilities for the existing loop configuration. Note in particular that the per-weld leak probabilities differ by up to one order of magnitude at 10 years, and by almost two orders of magnitude by the end of plant life. Note also that while the per-weld leak probabilities for riser and bypass piping behave similarly over time, the larger number of riser welds (20 compared to 10) and their somewhat higher per-weld leak probability are reflected in their dominant overall contribution to the probability of system leak (Fig. 12).

Figure 14 compares riser per-weld leak probabilities for 304SS and 316NG piping, in both cases based on the original loop configuration. Note the probability of leak in the 304SS weldment exceeds 1E-4 after only about 3 years of operation, while in Type 316NG this threshold is crossed only after some 15 years. The reason for this difference is clear from Fig. 15, which shows the total number of riser crack initiations in our evaluation (one weld, 25000 Monte Carlo replications) in both the 304SS and 316NG materials. Note that cracks initiate in 304SS within the first year of operation; by the time the first initiation occurs in the 316NG (about four years), nearly 1000 cracks have initiated in the less resistant material. The ratio of 316NG initiations to 304SS initiations falls to less than 100 at ten years, and to less than five by the end of plant life (see Fig. 16). By this time, however, piping in an actual plant would have gone through one or more ISI cycles.

Although the results presented here are only for the representative riser weld (i.e. the dominant contributor to the probability of system failure), they are characteristic of what we observed for the other welds considered. In all cases, the 316NG appears to owe its corrosion resistance mainly to the fact that (1) fewer cracks initiated than in the 304SS material, and (2) those that did initiate typically did so later in plant life. Once a crack initiates, however, its subsequent growth rate is not significantly affected by material type.

6. SUMMARY AND CONCLUSIONS

6.1 Discussion of Results

As part of our evaluations of reactor coolant piping for the Nuclear Regulatory Commission, we developed an advanced probabilistic model of stress corrosion cracking which we applied to the recirculation loops of a Mark I BWR plant. Based on the results of these evaluations, we were able to make the following general observations:

- if stress corrosion is not a factor, thermal fatigue is the main cause of pipe failure. Furthermore, the probability of break is similar to that in PWR reactor coolant loop piping (on the order of 1E-10 per reactor-year or lower). As for PWR reactor coolant loop piping, earthquakes contribute only negligibly to the probability of direct DEGB.

- when stress corrosion is a factor, corrosion-induced failure clearly dominates. Furthermore, the probability of pipe failure is dominated by residual stresses (i.e. uniform loads) rather than by stresses induced by applied loads. Our analyses further indicated that failure probability is very sensitive to the particular description of residual stress assumed in the analysis. This result may offer insight into field observations where nominally identical recirculation loops (e.g., in terms of configuration, materials, applied loads) may exhibit stress corrosion cracking in one plant and not in another. Such differences may be at least partly attributable to plant-to-plant differences in residual stresses caused by welding and "fit up" during pipe assembly.

- recirculation loops fabricated from Type 304 stainless steel are predicted to leak after about 10 years of operation. Although this result is based on conservative "worst case" stress assumptions and on the assumption of no in-service inspection over this period, it is also consistent with some field observations.

  If the 304SS material is replaced by 316NG and the existing loop configuration is retained, the system leak probability at ten years (a "typical" ISI interval) is effectively zero. The end-of-life system leak probability (i.e. after another 30 years of operation) is about 5E-1 per loop, or about 2E-2 per loop-year assuming "worst case" applied stresses and no ISI.

- for recirculation loops fabricated from Type 304 stainless steel, the system probability of DEGB is about 1E-2 after ten years of operation (or about 1E-3 per loop-year), increasing to about 2E-2 by the end of plant life. Again, these results reflect "worst case" applied stresses and no ISI.

- for recirculation loops fabricated from Type 316NG stainless steel, the system probability of DEGB is zero for the first 30 years of operation, even under "worst case" applied stresses and no ISI. In our evaluation we predicted only two riser breaks (out of 25000 Monte Carlo replications), none in other weld types, which implies a per-loop DEGB probability on the order of 1E-4 per year or less over the final ten years of plant life, zero up to that time. Routine ISI over plant life could be expected to substantially lower the "late-life" probability of DEGB though early detection of potentially troublesome cracks.

  Note that for 316NG, our "intergranular" stress corrosion model was actually based on laboratory data for transgranular stress corrosion cracking; we were unable to find suitable IGSCC data. Consequently, we would expect corrosion-induced cracking to more likely be transgranular rather than intergranular, and the probability of failure induced by "IGSCC" to actually be less than implied by our evaluations.

- for the replacement Type 316NG loop configuration, comprising fewer welds (about 30 compared to 50) and eliminating the bypass line altogether, the end-of-life leak and break probabilities drop by about a factor of two. Interestingly, the time-dependence of the system leak and break probabilities does not change significantly, reflecting the observation that the risers, rather than the bypass piping, dominate the probability of system failure.

- where failure in Type 304 piping is always dominated by initiated cracks (i.e., resulting from stress corrosion), in 316NG the initiated cracks dominate the probability of leak only after about 12 years. Once cracks are present, growth rates are nominally the same in either material. Consequently, the predicted difference in behavior between the two materials is due to differences in the number of initiated cracks and their later times-to-initiation, rather than how these cracks would grow once initiated.

6.2 Current and Future Applications

The NRC Office of Nuclear Reactor Regulation (NRR) recently published NUREG-0313, Rev. 2, which describes methods acceptable for controlling the susceptibility of BWR reactor coolant piping to intergranular stress corrosion cracking [11]. Although the NRR staff prefers replacement of sensitive piping with piping fabricated from IGSCC-resistant materials such as Type 316NG, enhancement of existing piping by appropriate combinations of repair (e.g., weld overlay, IHSI), prevention (e.g., hydrogen water chemistry), and augmented ISI is also an acceptable option for plant licensees. For example, the NRR guidelines specify various inspection intervals and sample sizes, depending on IGSCC mitigating measures that have been applied to an affected piping system, but do not define the specific welds that must be inspected.

The results of our recirculation loop evaluation indicated that the likelihood of pipe failure (i.e. leak or break) can vary widely among the weld joints in a piping system. Consequently, the specific welds selected at any given inspection could have a significant influence on system safety. As part of a new project for the NRC Office of Nuclear Regulatory Research, we are using the PRAISE computer code, and in particular our probabilistic model of stress corrosion cracking, to establish an inspection priority for BWR recirculation loop welds on the basis of calculated leak rates for the "representative" Mark I BWR plant in our earlier evaluation. Although not intended in itself to define an "acceptable" piping inspection program, it will provide NRR with one technical basis for reviewing utility responses to NUREG-0313, Revision 2.

The usefulness of probabilistic evaluations in regulatory applications has already been demonstrated through recent NRC rulemaking actions based in large part on the results of LLNL piping reliability studies. Although not a part of our present work, future licensing assessments related to the issue of stress corrosion cracking might conceivably include the following:

- development of specific licensing criteria. It is presumed that the criteria now included in NUREG-0313, Revision 2, will provide the basis for future NRR licensing decisions pertaining to BWR piping susceptible to IGSCC. Probabilistic evaluations like the one discussed in this paper could conveniently be applied to more fundamentally define just what constitutes an "acceptable" piping inspection program.

- assessment of the effectiveness of the recommended inspection schedules relative to alternate inspection schemes (e.g. more or less frequent inspection, greater or lesser extent of inspection).

- assessment of the effectiveness, either relative or absolute, of various measures for enhancing the performance of piping susceptible to stress corrosion cracking.

In principle, our probabilistic approach could be applied without modification to the first two of these activities, although additional work to improve PRAISE code efficiency would be desirable. The approach could also be applied to the third given appropriate PRAISE code modifications, such as the capability to change residual stress patterns, coolant conditions, and pipe geometry at selected times during plant life to model, respectively, IHSI, hydrogen water chemistry, and weld overlay. Such capability would be a powerful tool for future licensing assessment and shoud be considered for further development and application.

REFERENCES

1. **Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant**, Lawrence Livermore National Laboratory, Report UCID-18967, NUREG/CR-2189, Vol. 1-9 (1981).

2. **Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants**, Lawrence Livermore National Laboratory, Report UCID-19988, NUREG/CR-3660, Vol. 1-4 (1984).

3. **Probability of Pipe Failure in the Reactor Coolant Loops of Combustion Engineering PWR Plants**, Lawrence Livermore National Laboratory, Report UCRL-53500, NUREG/CR-3663, Vol. 1-3 (1984).

4. **Probability of Pipe Failure in the Reactor Coolant Loops of Babcock & Wilcox PWR Plants**, Lawrence Livermore National Laboratory, Report UCID-53644, NUREG/CR-4290, Vol. 1-2 (1985).

5. **Probability of Failure in the Reactor Coolant Piping of BWR Plants**, Lawrence Livermore National Laboratory, Report UCID-20914, NUREG/CR-4792, Vol. 1-4 (1986/88).

6. G.S. Holman and T. Lo, "Guillotine Breaks Indirectly Caused by Seismically-Induced Failures", Lawrence Livermore National Laboratory, Report UCRL-98172 (May 1988). Presented at the NRC-MITI Technical Information Exchange Engineering Workshop, Tokyo, Japan, May 16-20, 1988.

7. D.O. Harris, et al., **Probabilistic Fracture Mechanics Models Developed for Piping Reliability Assessment in Light Water Reactors**, Lawrence Livermore National Laboratory, Report UCRL-15490, NUREG/CR-2301 (April 1982).

8. T. Lo, et al., "Failure Probability of PWR Reactor Coolant Loop Piping," Lawrence Livermore National Laboratory, Report UCRL-86249 (February 1984).

9. **Probability of Failure in BWR Reactor Coolant Piping, Vol. 3: Probabilistic Treatment of Stress Corrosion Cracking in 304 and 316NG BWR Piping Weldments**, Lawrence Livermore National Laboratory, Report UCID-20914, NUREG/CR-4792, Vol. 3 (November 1986).

10. W.J. Shack, "Effects of Impurities on Stress Corrosion Cracking in BWR Piping Systems", Argonne National Laboratory (May 1988). Presented at the NRC-MITI Technical Information Exchange Engineering Workshop, Tokyo, Japan, May 16-20, 1988.

11. W. Hazelton and W. Koo, **Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping**, U.S. Nuclear Regulatory Commission, NUREG-0313, Revision 2 (1987).

Table 1. Pipe diameters, number of welds in existing and proposed recirculation loop configurations for BWR pilot plant.

Weld Group	Diameter (in)	Welds/loop (existing)	Welds/loop (proposed)
Discharge	26	10	11
Suction	26	6	5
Header	20	5	2
Risers	12	20	12
Bypass	3	10	0
Total per loop		51	30

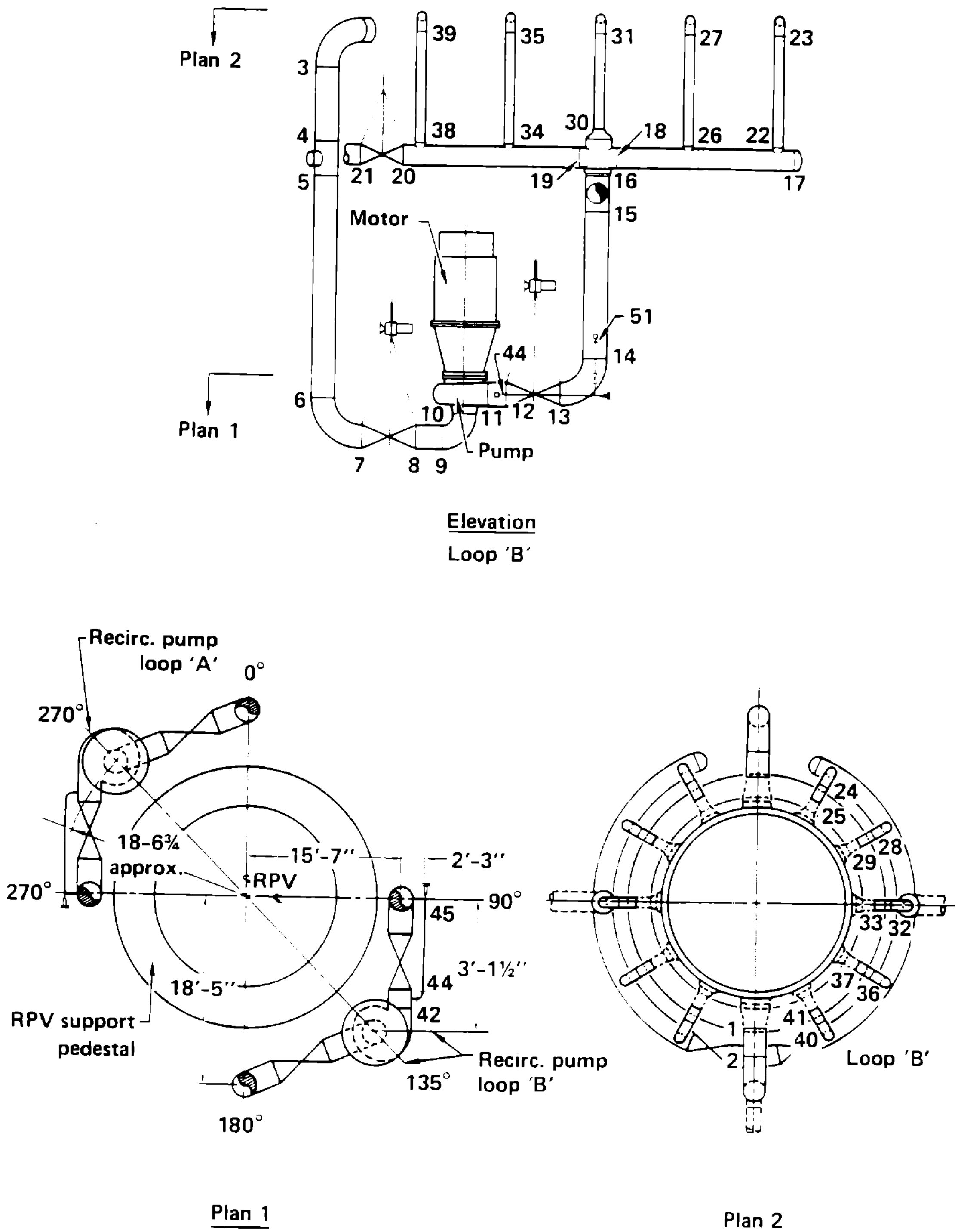


Fig. 1. Pilot plant recirculation system (existing configuration).

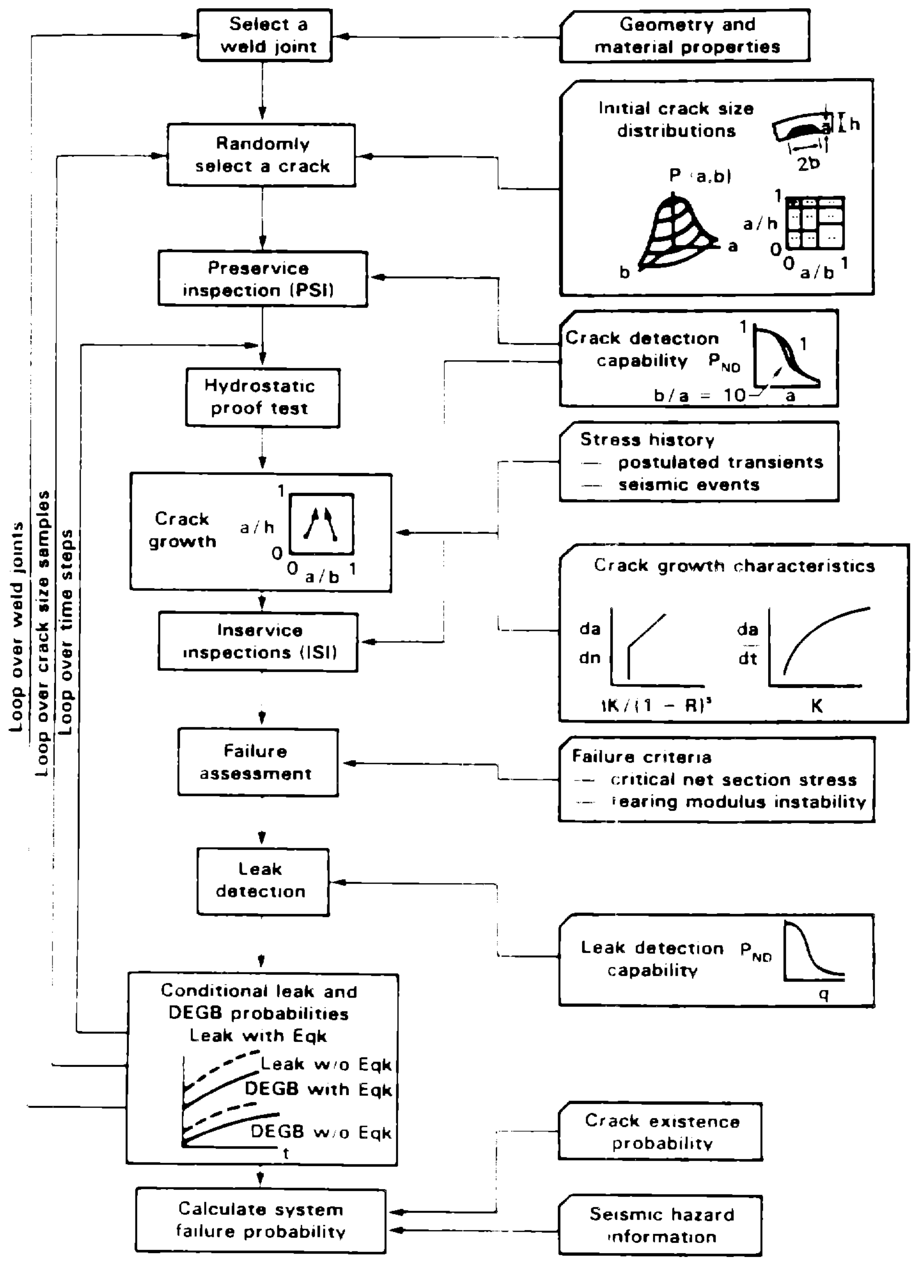


Fig. 2. Flowchart of the probabilistic fracture mechanics model implemented in the PRAISE computer code.

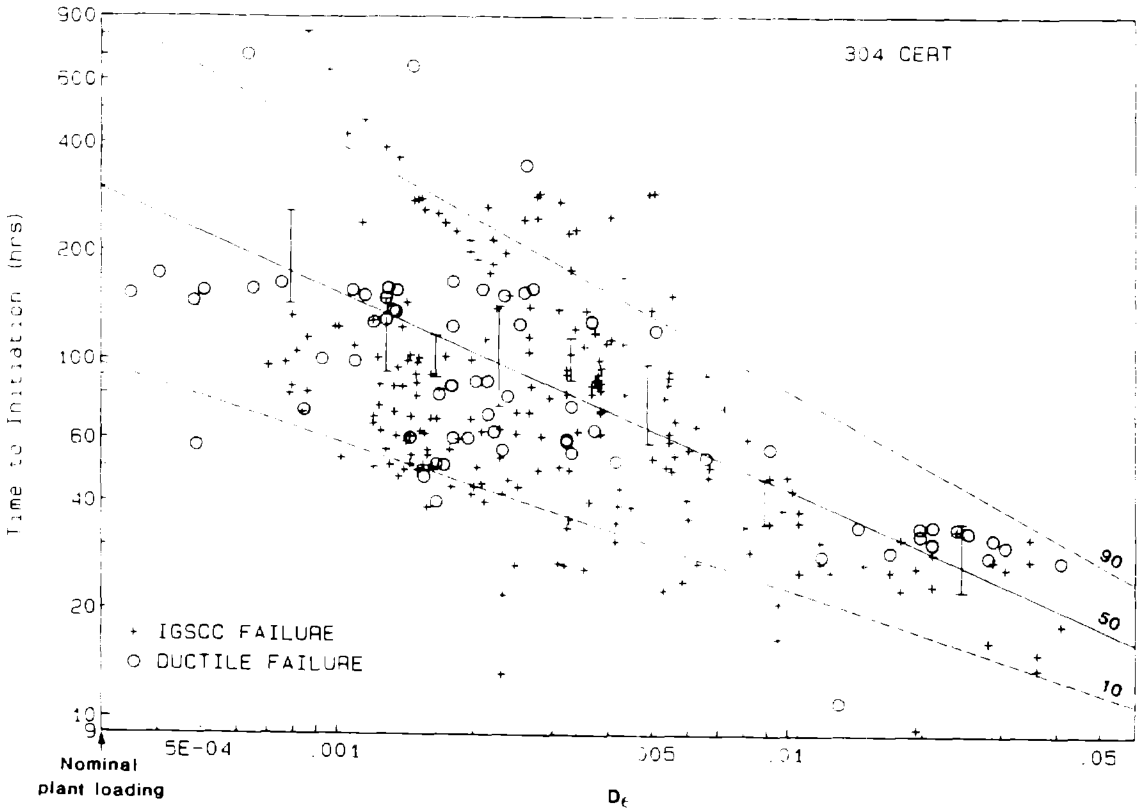


Fig. 3. Time-to-initiation for IGSCC cracks in 304SS as a function of damage parameter, plant loading/unloading.

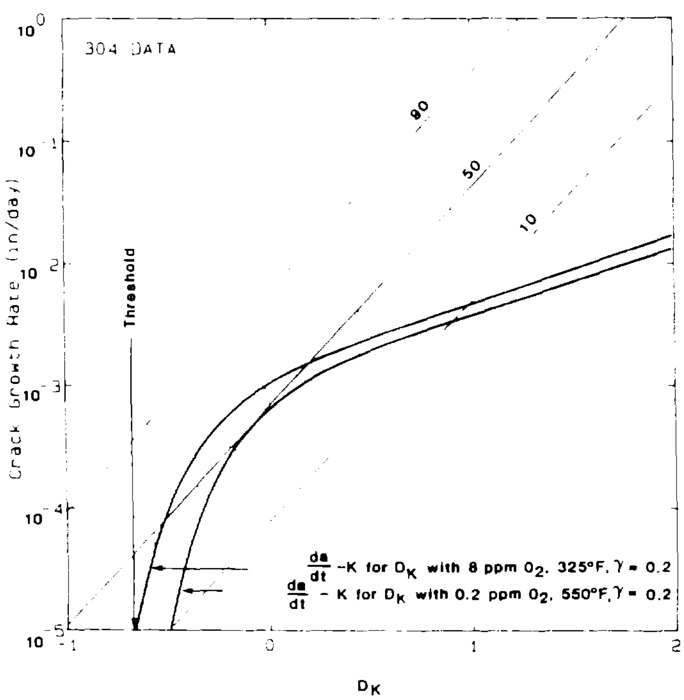


Fig. 4. IGSCC crack growth rate in 304SS as a function of damage parameter, steady-state operation. The solid lines represent crack growth rates predicted by the earlier IGSCC model in PRAISE.

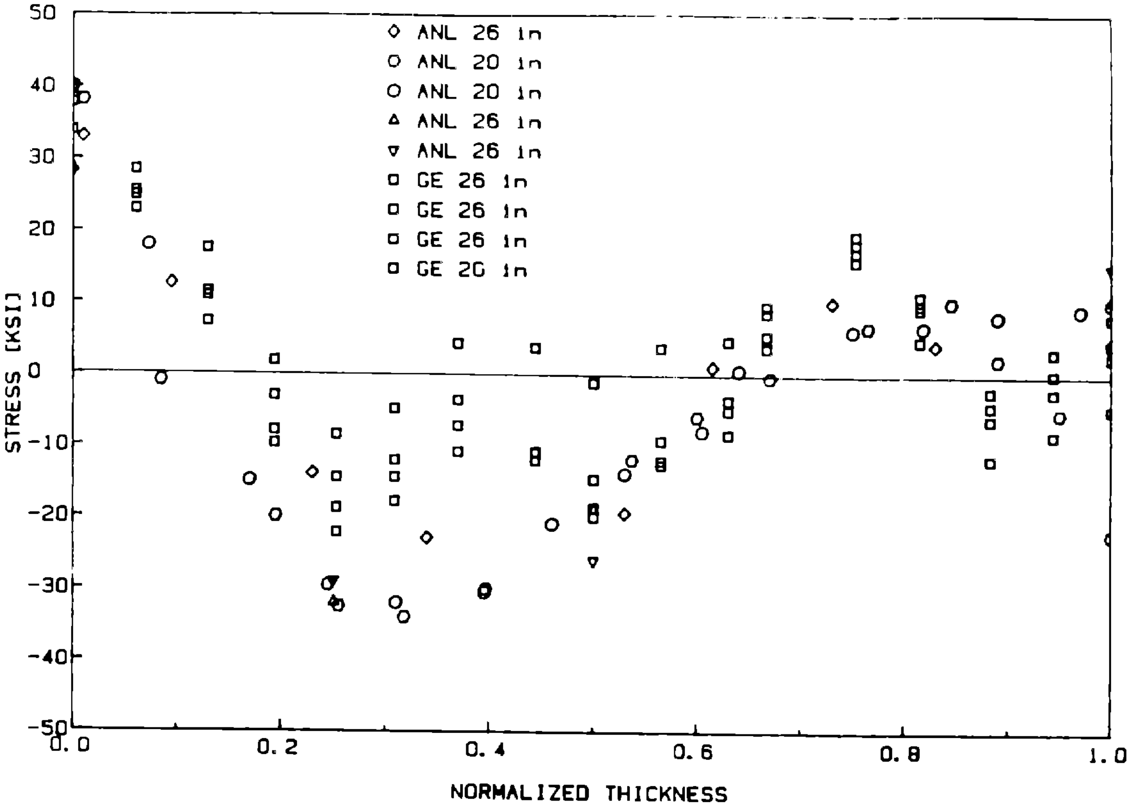


Fig. 5. Residual stress data for large-diameter (20 to 26 inches) piping as a function of distance from the inner wall.

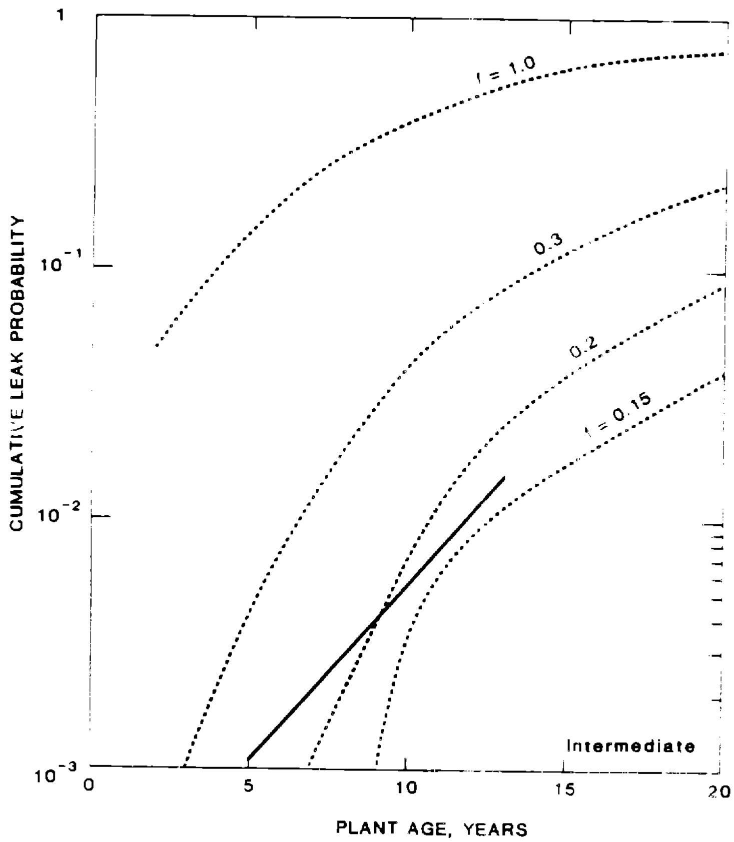
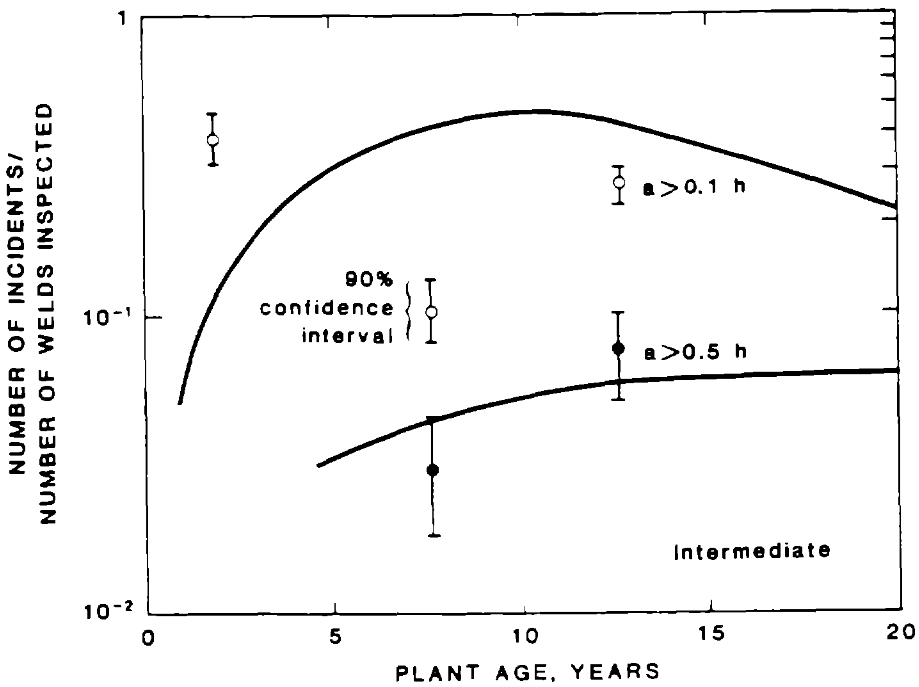


Fig. 6. Comparison of leak probabilities derived from field data with leak probabilities estimated by PRAISE for various values of residual stress adjustment factor (304SS).



●, o field data (a = crack depth, h = wall thickness) with 90% confidence interval

—— crack size distributions estimated by PRAISE for a > 0.1h (top) and a > 0.5h

Fig. 7. Comparison of crack indications derived from field data with PRAISE results based on optimum value of residual stress adjustment factor (304SS).

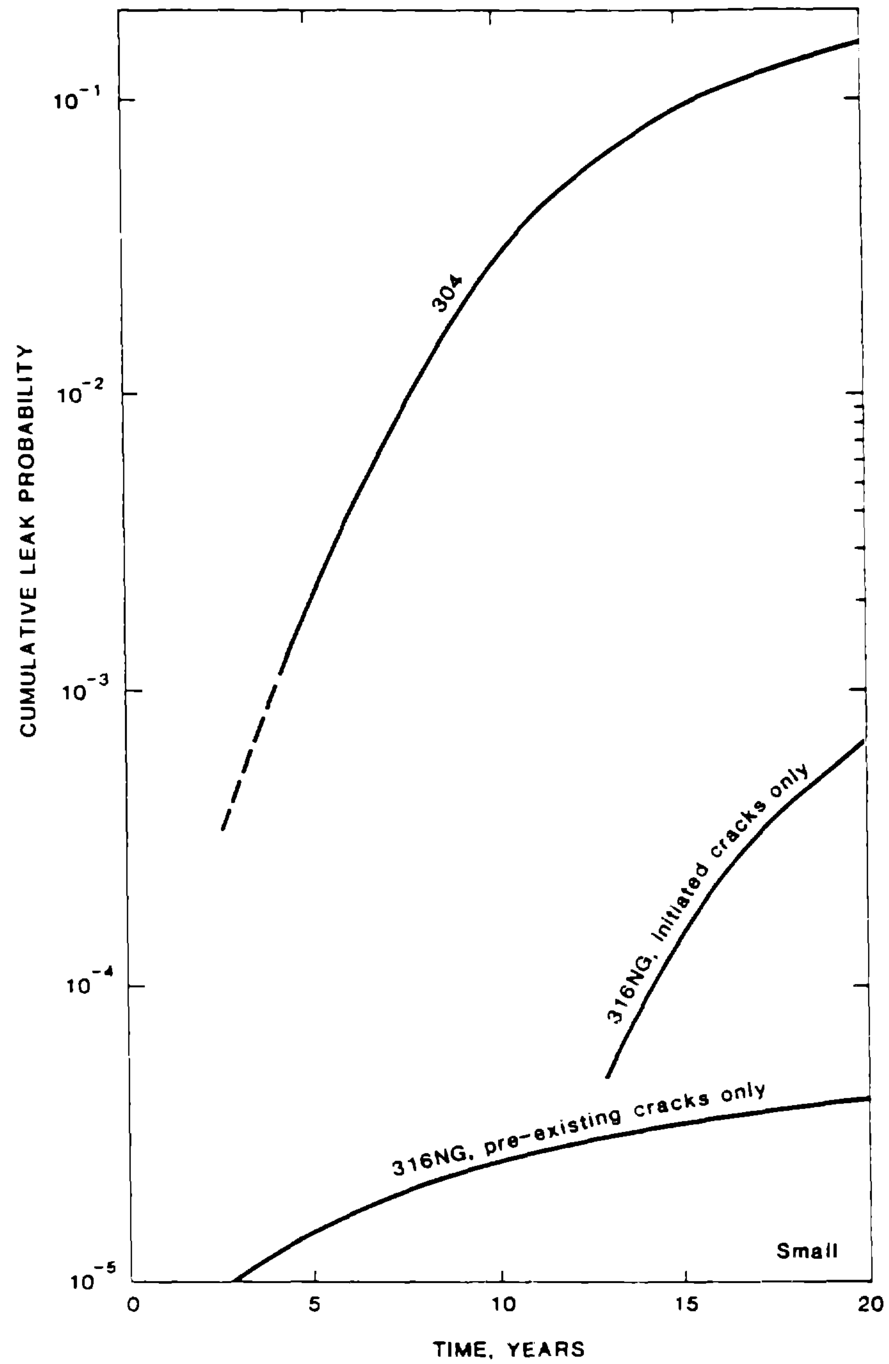


Fig. 8. Cumulative leak probability as a function of time for small-diameter weldments fabricated from Types 304 and 316NG stainless steel.

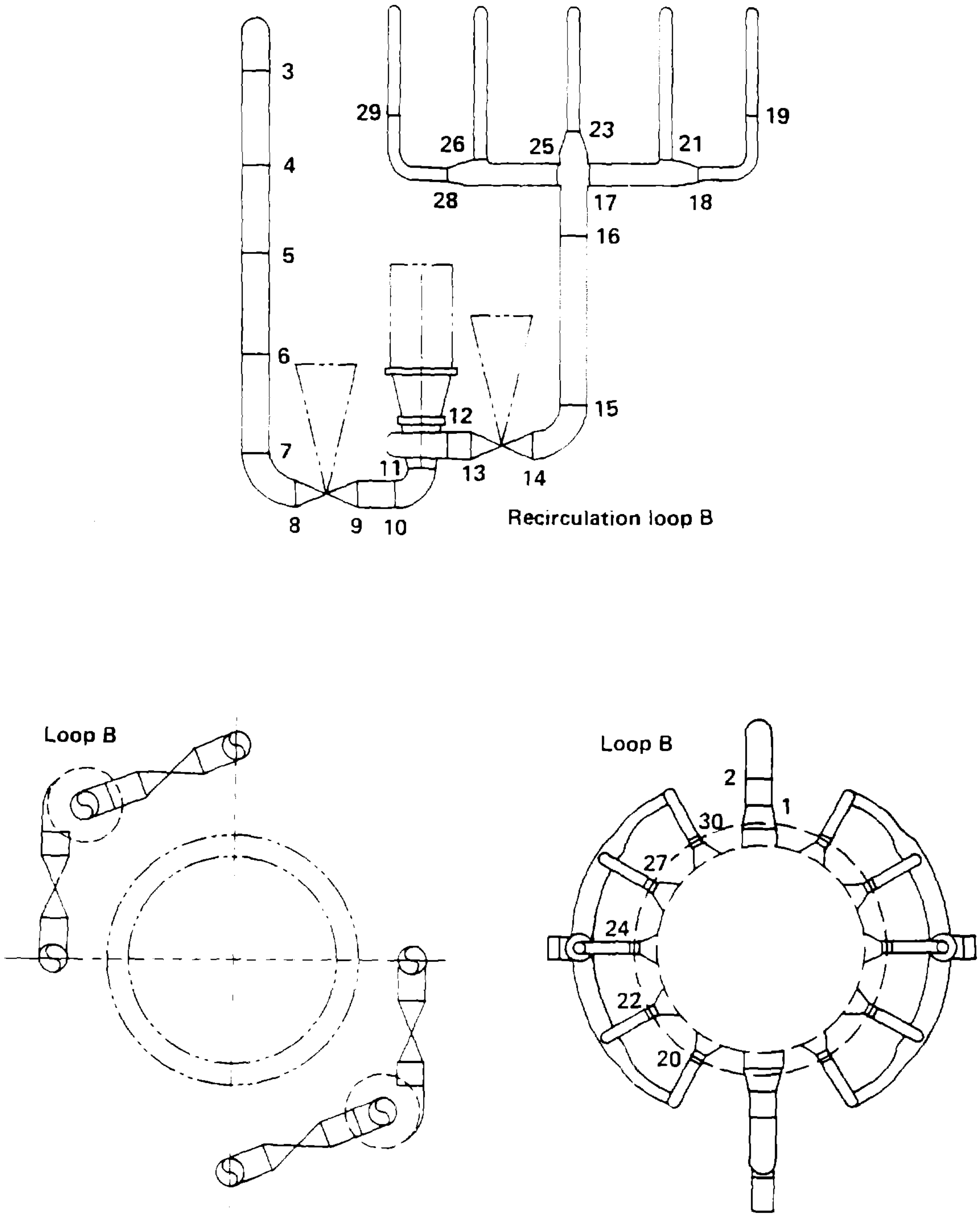


Fig. 9. Replacement recirculation loop configuration proposed for BWR pilot plant.

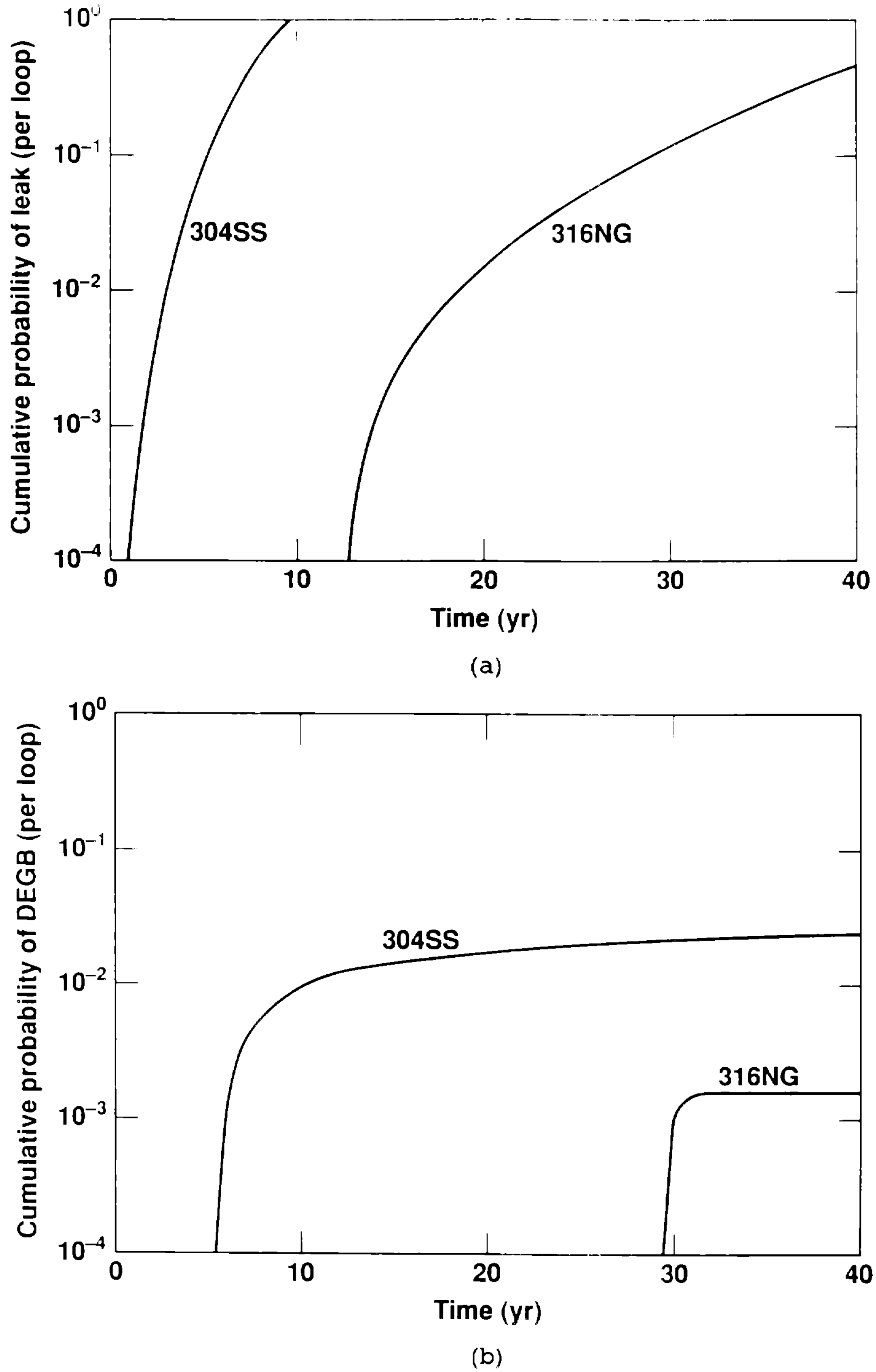


Fig. 10. Cumulative system probabilities of (a) leak and (b) DEGB for one pilot plant recirculation loop (existing configuration).

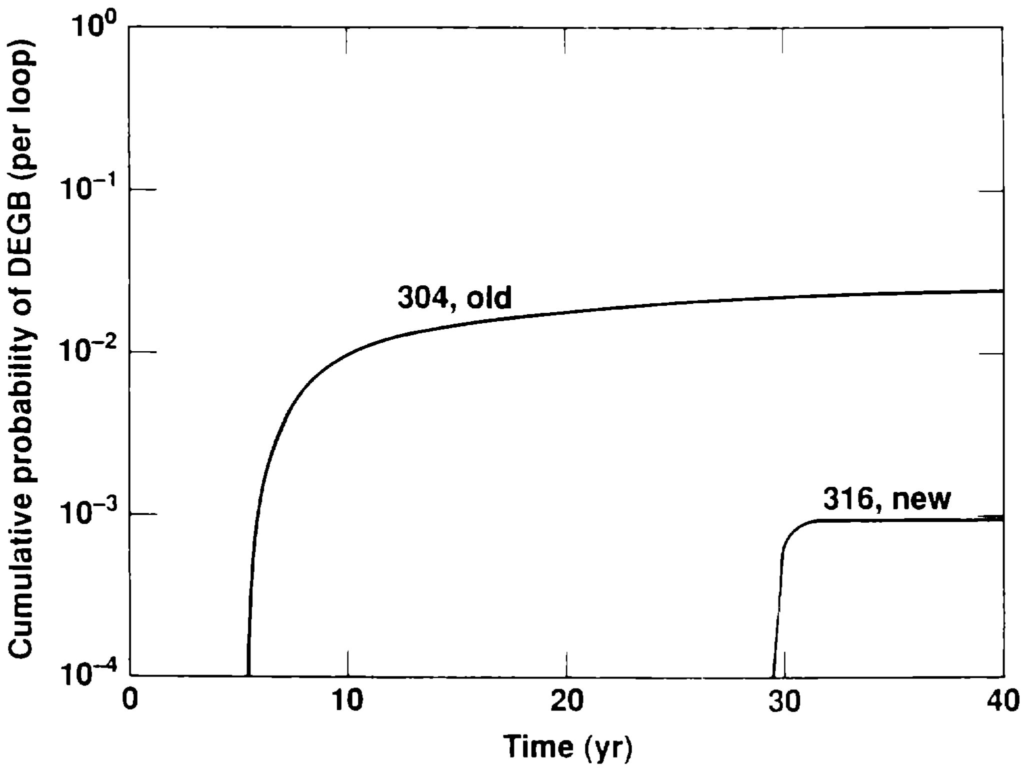
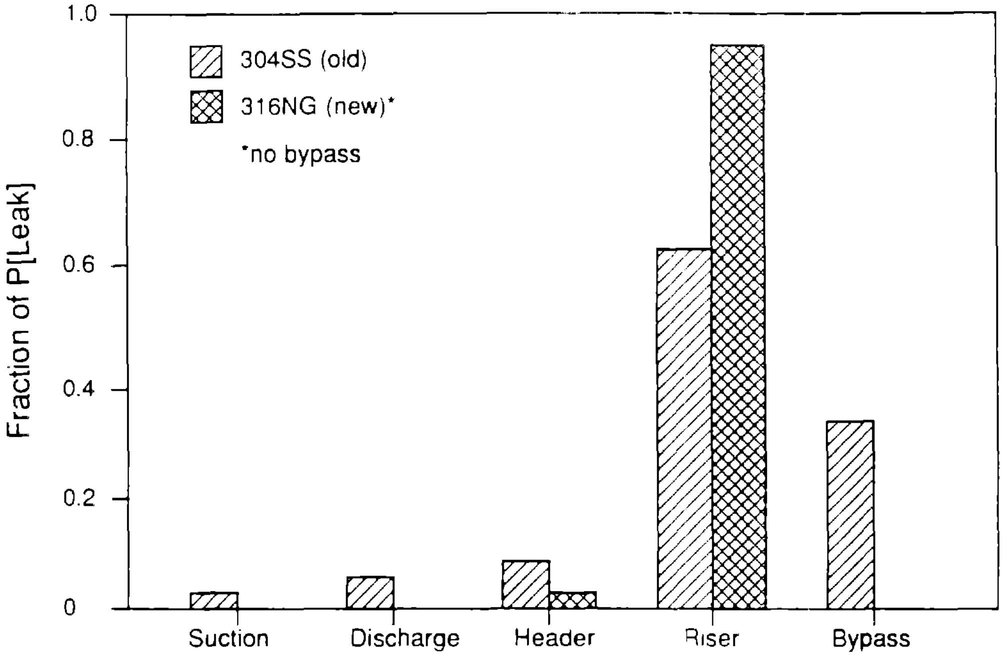
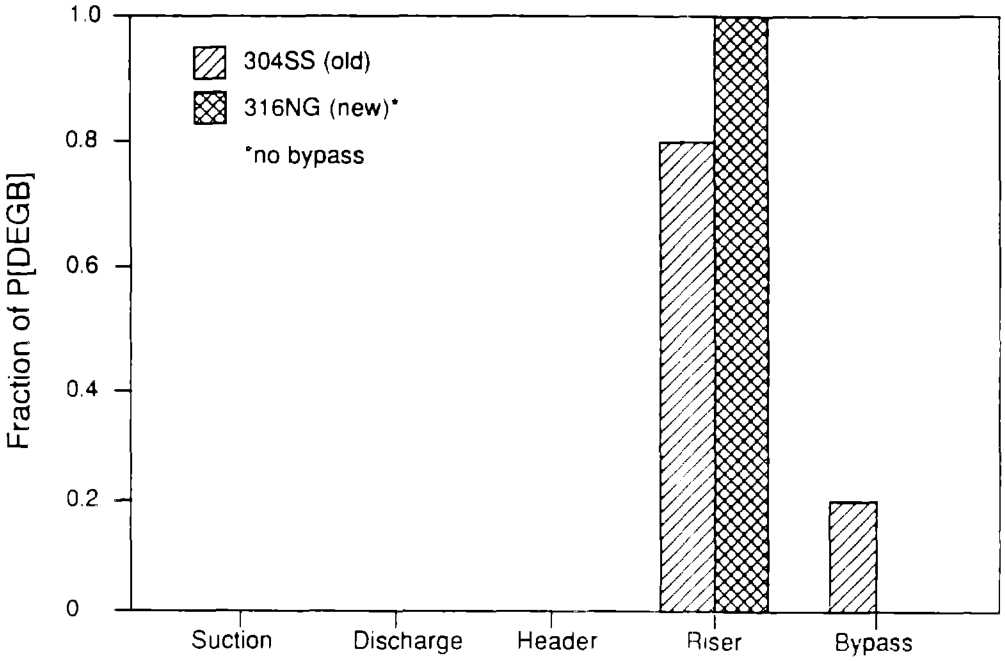


Fig. 11. Comparison of cumulative DEGB probabilities between existing recirculation loop and proposed replacement configuration.



(a)



(b)

Fig. 12. Relative contribution of various weld types to system probabilities of (a) leak and (b) DEGB, existing recirculation loops and proposed replacement configuration.

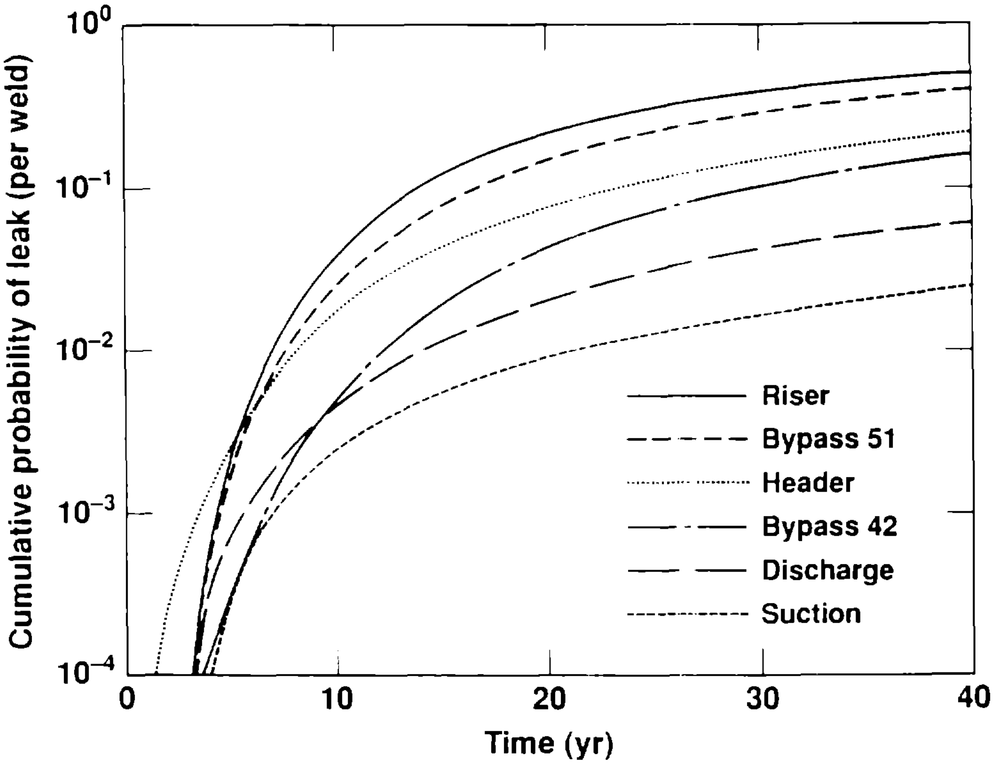


Fig. 13. Cumulative probabilities of leak for indicated welds (existing loop configuration).

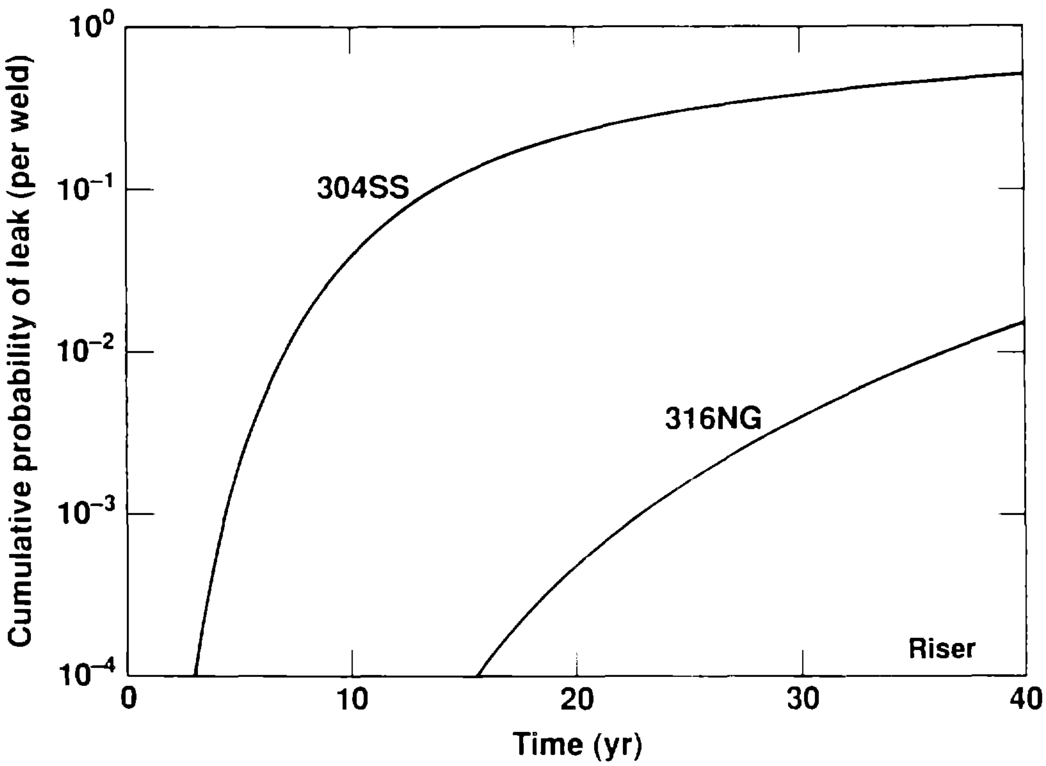


Fig. 14. Riser weld cumulative probability of leak, for Type 304 and Type 316NG stainless steel (existing loop configuration).

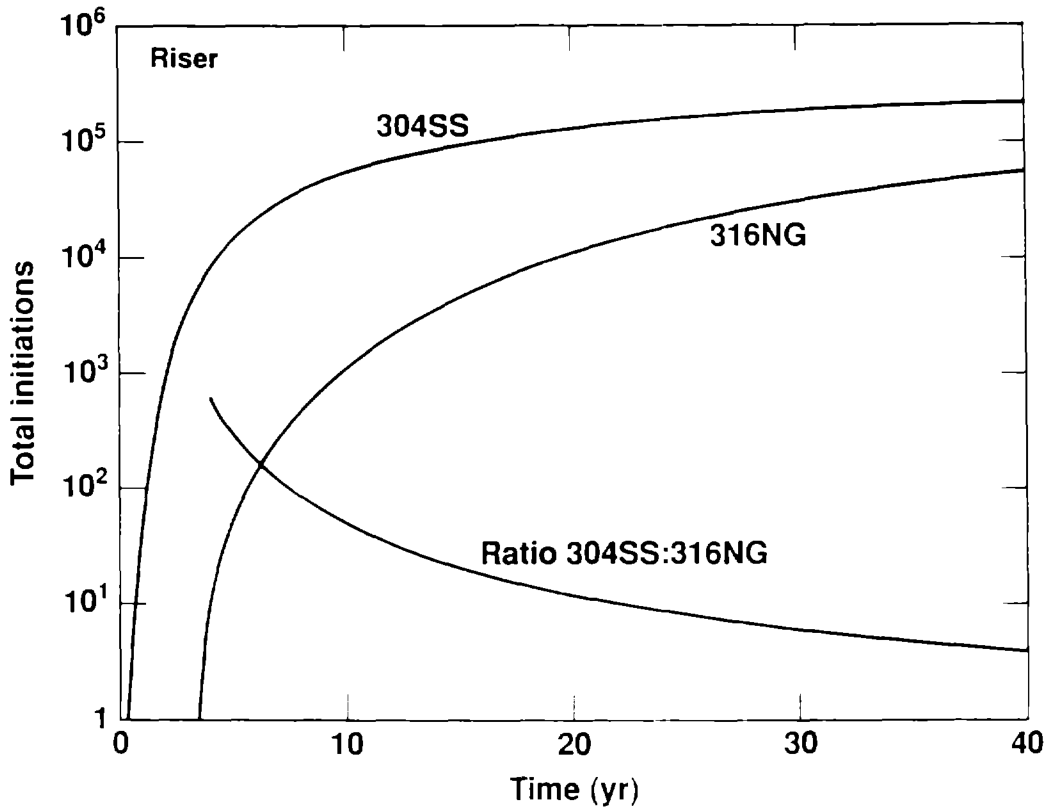


Fig. 15. Total riser weld initiations for Type 304 and Type 316NG material, existing loop configuration (25,000 Monte Carlo replications).

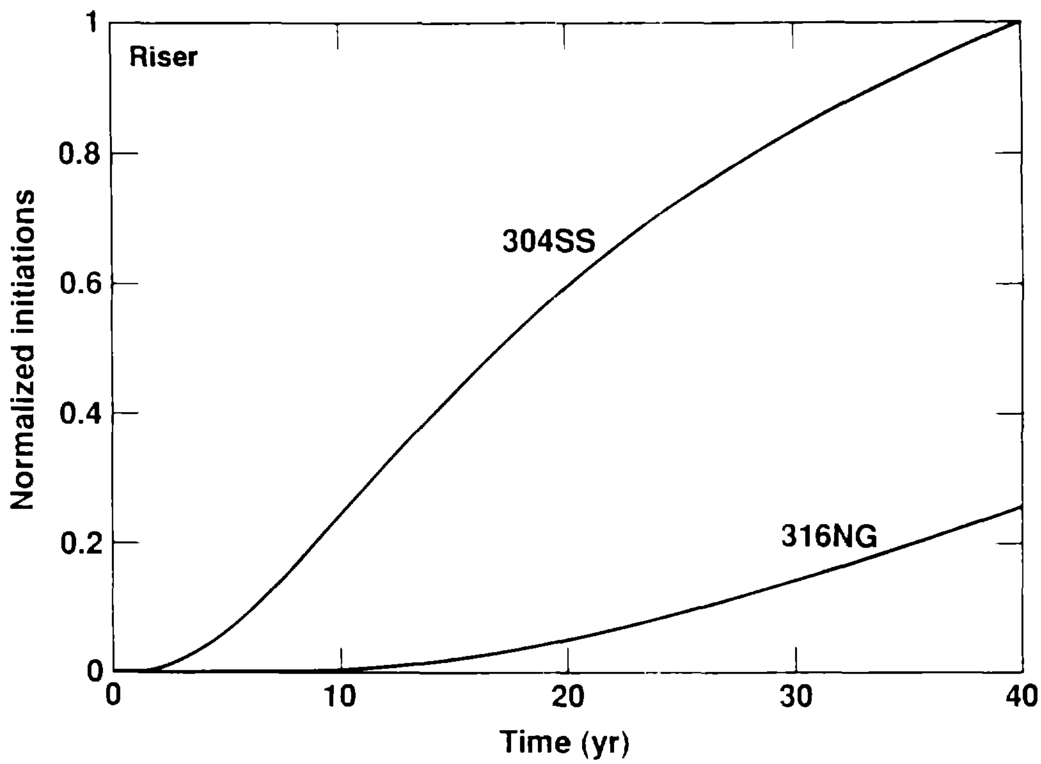


Fig. 16. Cumulative distribution of riser weld initiated cracks, normalized to Type 304 lifetime total.